Figure 4-14. Effects of Alternative Conceptual Models at 10,000 Years (Mohanty, et al., 2002, Figure 4-15)

rockfall rubble may create a barrier against potential magma flow down the drifts, based on analogy with channelized flow of lavas. The extent of blockage necessary to halt a pressure-driven flow is uncertain at this time. In addition, partial backfill of waste packages and drip shields by rubble may create a buffer zone between possible magma and engineered materials, changing potential thermo-mechanical effects from conductive to convective. The possible rubble zone, however, may have elevated temperatures and advect corrosive gasses evolving from the potential magmas (Connor, et al., 1997). Effects on long-term barrier materials from these processes remain uncertain at this time. See Section 4.3.10.1 for more detailed discussion on the impacts of these uncertainties on magma intrusion into the drifts.

Dynamic Effects of Rockfall on Engineered Barriers: Low Significance to Waste Isolation

The mechanical response of the engineered barrier system to discrete dynamic rock-block impacts is dependent upon the formation of discrete rock blocks of sufficient size within rock units of the repository horizon. Rock blocks of sufficient size to damage engineered barrier system components are only expected for a small portion of the repository.

<u>Discussion</u>

It has been determined that the formation of discrete rock blocks of significant size within the lower lithophysal rock unit is unlikely because of its highly fractured nature (Gute, et al., 2003). The analysis of the middle nonlithophysal rock unit, however, indicated that there are rock blocks of sufficient size to cause damage to the drip shield.

Because the middle nonlithophysal rock unit represents a relatively small percentage of the repository footprint (i.e., less than 30 percent) and only 40 percent of the rock blocks in this unit will have a mass greater than 2.25 metric tons [2.48 tons], the potential effect of discrete dynamic rock block impacts on overall repository performance has been determined to be a low-risk scenario for mechanical disruption (Gute, et al., 2003).

<u>Uncertainties</u>

The primary uncertainties associated with assessing the potential effects of dynamic rockfall on the drip shield are the rock-block impact location and impact-velocity distributions. Assuming the rock-block impacts occur at the same location on the drip shield (i.e., the drip-shield crown) with varying impact velocities (corresponding to the calculated fall heights), it was determined that the damage incurred by the drip shields from this potential disruptive event is negligible (Gute, et al., 2003).

Effects of Seismic Loading on Engineered Barriers: Medium Significance to Waste Isolation

The mechanical response of the engineered barrier system and waste form to seismic loading is strongly affected by the extent of accumulated rock fall rubble and associated elevated temperatures.

<u>Discussion</u>

The drip shield and waste package may be breached by the accumulation of damage caused by multiple seismic loading events (in addition to other mechanically disruptive events). It is expected that seismic events will increase the effective static load because of rockfall on the drip shield and waste packages.

<u>Uncertainties</u>

At the present time, there is a great deal of uncertainty associated with the seismic threshold loads needed to generate appreciable drip-shield and waste package mechanical damage. These threshold loads are strongly affected by the anticipated in-drift conditions (i.e., the presence of accumulated rockfall rubble and the concomitant elevated temperatures) and the different levels of material and structural degradation that are caused by various corrosion processes, fabrication effects (residual stresses and loss of ductility from welding and quenching), and hydrogen embrittlement.

Effects of Faulting on Engineered Barriers: Low Significance to Waste Isolation

Mechanical disruption of the engineered barrier system caused by faulting is dependent upon the mechanical loads placed on the engineered barrier system components by fault

displacements. Analyses considering patterns of surface-faulting displacements from historical earthquakes in the Basin and Range physiographic province indicate the potential effect of direct fault rupture of engineered barrier system components is expected to be small.

Discussion

Analysis of the potential effect of direct fault rupture on the engineered barrier estimates the risk is small [on the order of tenths of microsieverts (tens of microrems) or less]. The significance of faulting has been evaluated by using the FAULTO module in the TPA Version 4.1j code and by considering the patterns of surface-faulting displacements from historical earthquakes in the Basin and Range physiographic province (Stamatakos, et al., 2003). Both analyses relied on conservative, and in some cases, upperbound assumptions. For example, no credit was taken for the mechanical strength of the waste packages because of limited data and analyses on the mechanical behavior of waste packages during a faulting event. All waste packages intersected by faults were assumed to fail. Similarly, because of limited data on the amount of distributed and secondary fault displacements anticipated in the repository during an earthquake at Yucca Mountain, the faulting analyses conservatively assumed that all secondary or distributed faults were capable of failing waste packages, independent of the amount of fault displacement that might occur on the secondary or distributed faults. Nonetheless, risk estimates by both methods are similar and very small as shown in Figure 4-15. These analyses show that faulting does not contribute significantly to overall repository risk, even with the conservative and upperbound assumptions.
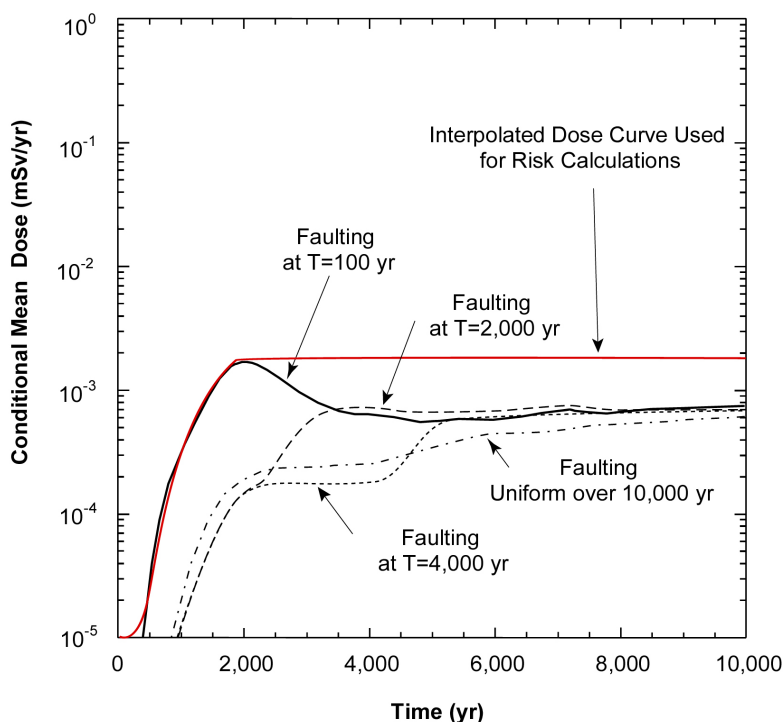

Figure 4-15. Comparison of the Mean Dose Versus Time Curves for Cases in which the Faulting Events on the Solitario Canyon Fault Were Forced to Occur in All 500 Realizations at a Specified Year (100, 2,000, and 4,000) and for a Case in Which the Faulting Events Occurred Uniformly over the Entire 10,000-Year Postclosure Period

4-33

There are large uncertainties in both the amount and likelihood of fault displacements, and the associated mechanical loads placed on the waste packages and drip shields by these displacements. However, the bounding assumptions employed in the analyses discussed above provide confidence that these uncertainties are still not expected to result in faulting significantly contributing to overall repository risk.

4.3.3 Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms (ENG3)

Risk Insights:
   Chemistry of Seepage Water   High Significance

4.3.3.1 Discussion of the Risk Insights

Chemistry of Seepage Water: High Significance to Waste Isolation

Evaluating the range in chemistry of water seeping into the drift and contacting the drip shield and waste package is important for determining corrosion rates of the engineered materials.

Discussion

The quantity and chemistry of water that drips onto drip-shield and waste-package surfaces largely determine the composition of the salts that can precipitate on those surfaces. The composition of a salt determines the composition of its associated brine, and the relative humidity conditions under which that salt will deliquesce to form a brine. The composition of the salt is largely determined by the chemistry of the evaporating water, whereas the timing and extent of salt formation/evaporation is largely determined by thermal-hydrological conditions in and above the repository over time. Changes in seepage rate and the conditions appropriate for evaporation and salt formation depend on diverse factors, such as climate, flow pathways in the unsaturated zone, heat load, drift design, and local controls on flow pathways (i.e., roughness of drift crown surface, the location of rock bolts and other engineered materials, and the timing and extent of rock fall and drift degradation).

The expected lifetimes of the drip shield and waste package depend strongly on the chemistry of water contacting their surfaces. Low pH and elevated concentrations of certain dissolved anionic species are expected to enhance aqueous corrosion—promoting specific corrosion modes, either uniform corrosion, or localized corrosion, the latter of which has orders of magnitude higher rates than uniform corrosion. For example, the susceptibility to localized corrosion of Alloy 22, the material DOE plans to use for the outer container of the waste package is strongly dependent on pH and on chloride ion concentration (Dunn, et al., 2003). Titanium Grade 7, the material planned to be used for drip shields that would cover the waste packages, has been shown to be susceptible to generalized corrosion from fluoride ions (Brossia, et al., 2001), and other data indicate that titanium is susceptible to localized corrosion

in the presence of bromide ions (Beck, 1971). On the other hand, other anionic species can inhibit localized corrosion if present in sufficient concentration. For example, nitrate ions have been shown to inhibit localized corrosion of Alloy 22 in chloride solutions (Dunn, et al., 2003). However, titanium corrosion may be limited by the supply of fluoride from dripping water and not strictly to the concentration threshold (Lin, et al., 2003).

High concentrations of these anionic species could result from the evaporation of seepage waters, even if those waters initially were dilute (Brossia, et al., 2001). Evaporation experiments (Rosenberg, et al., 2001) and some limited thermodynamic analyses (Pabalan, et al., 2002a,b) show that waters of widely varying initial composition evolve by evaporation toward three types of brines: (i) calcium-chloride (Ca-Cl), (ii) neutral, and (iii) alkaline brines. The results of thermodynamic simulations of evaporation of Yucca Mountain ground waters indicate that Ca-Cl brines tend to have chloride-to-nitrate ratios sufficiently high to make the corrosion-inhibiting property of nitrate ions ineffective (Pabalan, et al., 2002a,b). The thermodynamic calculations also show that alkaline brines tend to have fluoride ion concentrations that exceed 0.0005 moles/kg H_2O, a threshold fluoride concentration at whichTitanium Grade 7 has been observed to undergo accelerated general corrosion in deareated 1-M sodium chloride solutions (pH 6.4 to 8.2) at 95 °C [203 °F] (Brossia, et al., 2001). The thermodynamic analyses indicate that the three brine types have elevated bromide concentrations, but the potential effect of the elevated bromide concentrations is uncertain because of the limited data on corrosion of titanium in bromide solutions.

In addition to elevated concentrations of anionic species, evaporation of seepage waters possibly could lead to acidic condensates that could enhance the degradation of engineered materials. For example, experiments by Pulvirenti, et al. (2003) demonstrated a high corrosion rate for Alloy 22 that was reacted at 130 °C [266 °F] with acidic condensate formed by evaporation of simulated Yucca Mountain brines. The fast corrosion rate was interpreted as resulting from the formation of hydrochloric acid and nitric acid during evaporation of the highly concentrated brines. The formation of hydrochloric and nitric acid was supported by chemical tests of vapor samples in a closed system. While comparable temperatures are expected for the repository, the open nature of the Yucca Mountain system (i.e., the movement of air in the unsaturated zone), and the buffering effect due to minerals in the host rock, significantly limit the concentration of hydrochloric and nitric acid.

Additional salts, such as soluble chlorides, nitrates, and sulfates of sodium, potassium, calcium, and magnesium, that are present as aerosols in atmospheric air (Ge, et al., 1998) and entrained in ventilation air introduced into the repository drift also may be deposited on the drip-shield and waste-package surfaces (Bechtel SAIC Company, LLC, 2001). Inorganic salts are generally hygroscopic and will absorb moisture from humid air, generating small volumes of potentially corrosive brines. A phase change from a solid particle to a saturated aqueous phase usually occurs spontaneously when the relative humidity in the surrounding atmosphere increases to a level known as the deliquescence point or deliquescence relative humidity (Tang and Munkelwitz, 1993). The deliquescence relative humidity of salt mixtures depends on the mixture composition and temperature. Experimental data and thermodynamic analyses (Pabalan, et al., 2002a,b) show that mixtures containing magnesium and calcium have much lower deliquescence relative humidity than those containing sodium and potassium. If magnesium and calcium salts are deposited on waste package and drip-shield surfaces, deliquescence and

rine formation could occur when the repository temperature is above 100 °C [212 °F] and could lead to early initiation of aqueous corrosion of the drip shield and waste package engineered barrier.

Uncertainties

Some uncertainty arises from the uncertainty in the quantity of water entering the drift, and the timing of and the temperature during which seepage occurs. Published experimental and modeling studies of evaporation have implicitly assumed that seepage water is present and that the temperature is sufficiently high to cause evaporation. However, the flux of seepage water over time depends on diverse factors, such as climate, flow pathways in the unsaturated zone, heat load, drift design, and local controls on flow pathways (i.e., roughness of drift crown surface, the location of rock bolts and other engineered materials, and the timing and extent of rockfall and drift degradation).

The most significant source of uncertainty in determining the chemical environment for corrosion is the range of in-drift water compositions that may result from spatial and temporal variations in seepage water composition, the composition and amount of condensed water formed by cold-trap processes, and the extent of chemical interactions between these waters and engineered and natural materials. Coupled thermal-hydrological-chemical processes occurring in the rocks that overlie the proposed repository will largely determine the quantity and chemistry of water seeping into the drifts. Explicit and comprehensive evaluation of these coupled processes, however, generally requires the construction of reactive transport models (CRWMS M&O, 2001).

Reactive transport predictions have two substantial sources of uncertainty that are difficult to quantify (Browning, et al., 2003a,b): (i) code limitations, especially algorithms that reflect a representation of coupled thermal-hydrological-chemical processes; and (ii) uncertain parameter values. A comprehensive evaluation of the uncertainties associated with conceptual models and code limitations requires an assessment of how omitted processes and components affect model results. It is generally not feasible to quantify precisely the uncertainties associated with the myriad individual input parameters and constraints typically associated with reactive transport models, limiting assessments to analyzing the effects for a range of potential chemistries. Nevertheless, these uncertainties are constrained by the system characteristics (e.g., relatively homogeneous rock chemistry) and by the well-understood dynamics of evaporation (i.e., the chemical divide phenomenon results in a diversity of water compositions evolving to a few brines).

The composition of in-drift water is a significant factor in determining the types of salts that can form and the corrosiveness of the associated brine. Bounding the range of in-drift water compositions requires characterization of diverse in-drift features, events, and processes that may alter the chemical composition of seepage waters. As noted above, these features, events and processes include cold-trap processes that control the timing, locations, and compositions of condensed waters; the development of preferential flow pathways that may focus flow over drip-shield and waste-package surfaces; the chemical consequences of degradation of the engineered drifts; and the chemical evolution of in-drift waters that have interacted chemically with engineered materials. The impact of these features, events, and processes on the quantity and chemistry of in-drift water is believed to be of less importance.

Some uncertainty also remains with respect to the composition of dust that may be deposited on

the drip-shield and waste-package surfaces, and the extent to which chemical interactions between dust and in-drift waters may alter the chemical environment for corrosion. Chemical analyses by the U.S. Geological Survey (Peterman, et al., 2003) on a limited set of dust samples from the exploratory shaft facility indicate that a large proportion of the dust in the Exploratory Studies Facility comes from communication of the rock during tunnel construction and autogenous grinding of muck during haulage on the conveyor belt. However, the U.S. Geologic Survey data also indicate that salts of evaporated construction and native pore water are minor components of the dust. There are limited data to support extrapolation of the composition of dust sampled from the Exploratory Studies Facility to that of dust that may be encountered in the repository drift. There is also uncertainty with respect to the deliquescence behavior of salts mixed with rock dusts.

4.3.4 Radionuclide Release Rates and Solubility Limits (ENG4)

Risk Insights:

Waste-Form Degradation Rate	Medium Significance
Cladding Degradation	Medium Significance
Solubility Limits	Medium Significance
Mode of Release from Waste Package	Low Significance
Effect of Colloids on Waste Package Releases	Medium Significance
Invert Flow and Transport	Low Significance
Criticality	Low Significance

4.3.4.1 Discussion of the Risk Insights

Waste-Form Degradation Rate: Medium Significance to Waste Isolation

The dissolution rate of the waste form in an aqueous environment is important for all radionuclides. Uncertainty in the dissolution rate is large such that the time required to release radionuclides from the spent nuclear fuel matrix or vitrified-waste forms can vary from hundreds of years to hundreds of thousands of years. Water chemistry and temperature within the waste package could affect the degradation rate of the spent nuclear fuel and vitrified wastes. Corrosion of the internal metallic components of the waste package (e.g., fuel assembly baskets) could reduce pH, leading to higher dissolution rates from spent nuclear fuel and vitrified wastes.

Discussion

Several studies point to the sensitivity of dose to the dissolution rate of the spent nuclear fuel source material. The preexponential coefficient for the basecase spent nuclear fuel dissolution model (exponential dissolution Model 2) in the TPA code is one of the most sensitive parameters (Table 4-5) (Mohanty, et al., 2002). Furthermore, among the four alternative models for spent nuclear fuel degradation in TPA Version 4.1 (which have markedly different release rates), there is a clear correlation between release rate and dose (Figure 4-16). Dissolution rate

Table 4-5. Influential Parameters for the 10,000-Year Simulation Period from Sensitivity Analysis Studies		
No.	Parameter Name	Score
1	Areal average mean annual infiltration at start	7/7
2	Drip-shield failure time	7/7
3	Preexponential term for spent nuclear fuel dissolution (Model 2)	6/7
4	Subarea wet fraction	6/7
5	Waste package flow multiplication factor	6/7
6	Well pumping rate at 20-km [12.4-mi] receptor group	6/7
7	Fraction of condensate toward repository	4/7
8	Alluvium retardation coefficient for Ne-237	3/7
9	Distance to tuff-alluvium interface	3/7
10	Waste package initially defective fraction	3/7

is a relatively important determinant of repository performance, but the overall peak dose is not directly proportional to it, because other mechanisms like diffusion, advection, and solubility attenuate ultimate release rates from the engineered barrier. The previous DOE total system performance assessment models use fuel dissolution rates considerably larger than those used by NRC (CRWMS M&O, 2000b). Using more aggressive chemical conditions (Model 1) in place of the basecase (Model 2) leads to a 2-order of magnitude increase in release rate, yet, as shown in Figure 4-16, the increase in peak expected dose is only a factor of approximately 2.5. NRC staff expect the basecase (Model 2) to represent chemical conditions to which spent nuclear fuel could be exposed under anticipated repository conditions.

Uncertainties

The dissolution rate of spent nuclear fuel depends on temperature, the presence of oxygen, and the chemistry of in-package water. There are similar controls on the dissolution of vitrified waste, although it is not significantly affected by oxidation potential. Furthermore, the rate at which radionuclides can leave the waste package may be restricted not strictly by the dissolution rate of the fuel but by: (i) the amount of water coming into contact with the waste form and escaping the waste package, (ii) diffusion processes inside and outside the waste package, and (iii) solubility. Model 4 (schoepite dissolution model) recognizes that the spent nuclear fuel can degrade relatively quickly in the presence of water and oxygen, but that some radionuclides may be incorporated into secondary minerals formed during spent nuclear fuel degradation and released at a slower rate by subsequent dissolution of these less-soluble phases.

Cladding Degradation: Medium Significance to Waste Isolation

Zircaloy cladding exhibits extremely low uniform corrosion rates in aqueous environments and could delay substantially the release of radionuclides from commercial spent nuclear fuel if it remains intact. Performance assessments estimate a high correlation between dose and fraction of failed cladding. However, cladding is thin and not physically strong. Cladding failure can occur as a result of localized corrosion, stress corrosion cracking, and hydride reorientation, under a combination of adverse environmental and stress conditions. Cladding may also fail
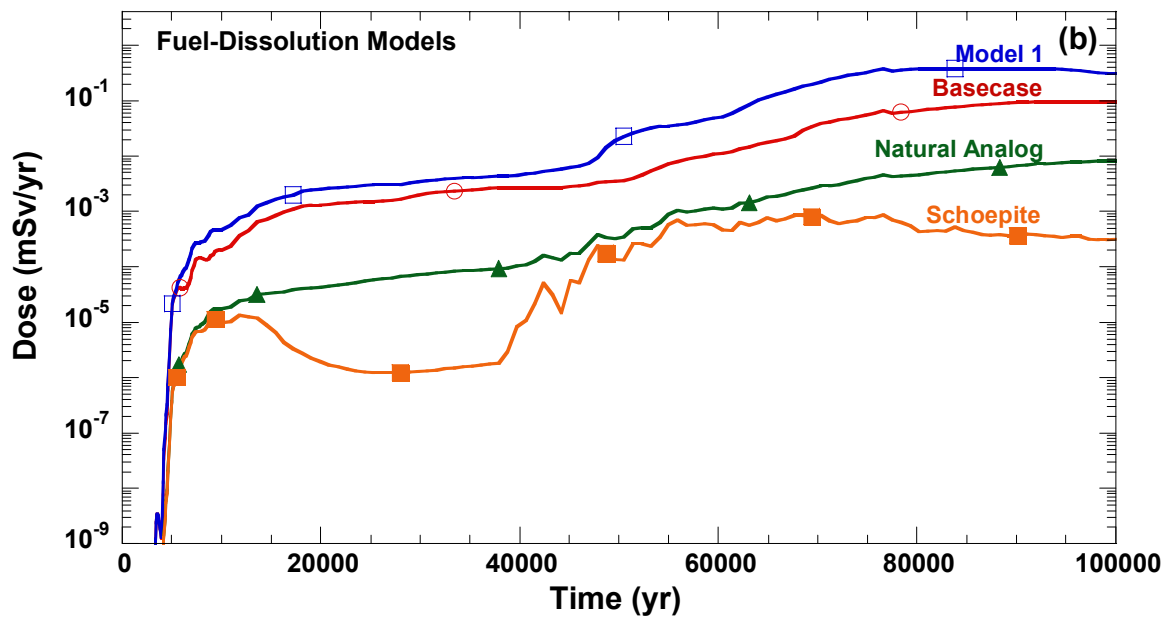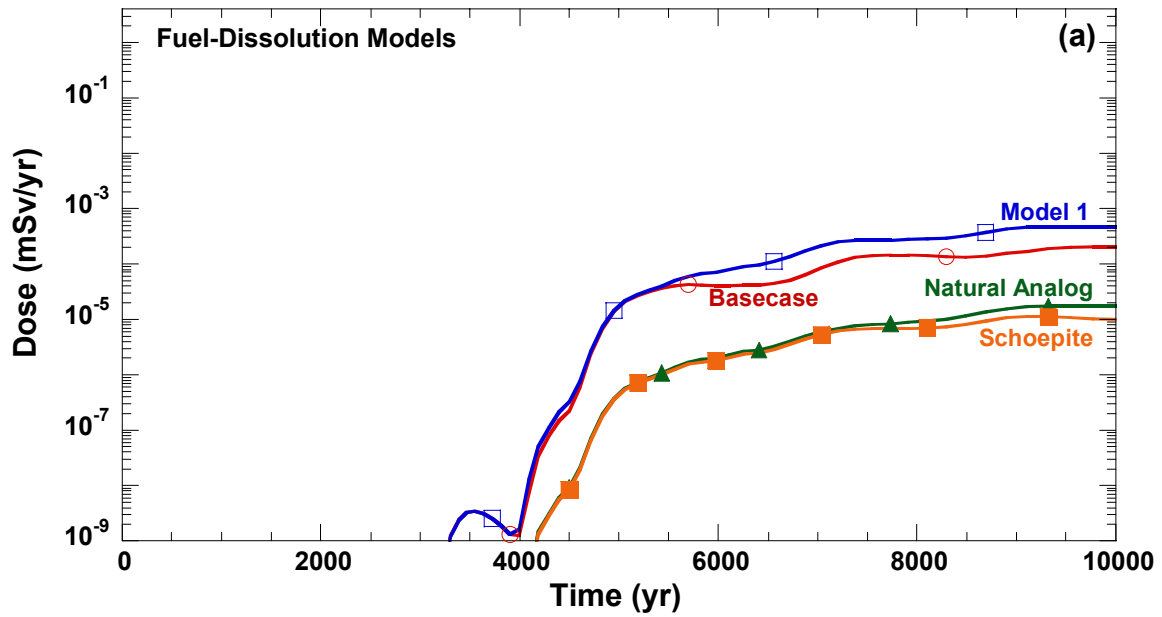
Figure 4-16. Ground Water Dose from the Basecase and the Fuel-Dissolution Alternative Conceptual Models for (a) 10,000 Years and (b) 100,000 Years, For 350 Realizations (Mohanty, et al., 2002, Figure 3-36)

as a result of creep caused by hoop stresses due to internal pressure, or mechanically, when subjected to loads associated with seismic events and rockfall.

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Discussion

DOE has considered that cladding can be an effective metallic barrier against the release of radionuclides from commercial spent nuclear fuel. However, little experimental evidence has been provided to support such assessment nor have solid technical bases for the assumptions included in the model abstraction been developed. This is the case, in particular, for the modeling of localized corrosion and stress corrosion cracking, as well as for the lack of consideration of hydride reorientation as a potential failure process that may be faster and, hence, more detrimental than unzipping alone. Recently, DOE has provided performance assessment calculations showing that the cladding degradation rate at the 95th percentile, or even complete neutralization, only increases the mean dose by one order of magnitude, and the dose is more than four orders of magnitude lower than that specified in the regulations at 10 CFR Part 63 for the reasonably maximally exposed individual (Figure 4-17). However, in their nominal case, DOE assumes that the fraction of failed cladding perforated before unzipping remains constant at 0.08 during the 10,000-year compliance period, reaching only 0.2 after 100,000 years (CRWMS M&O, 2000b). Also, these estimates of cladding protection do not consider the full range of possible failure mechanisms or their probabilities, and therefore, may overestimate the effectiveness of cladding as a barrier. In other international programs (e.g., France, Canada, Sweden, Finland), no credit is given to the cladding as a metallic barrier as a result of the uncertainties related to its integrity under repository conditions.
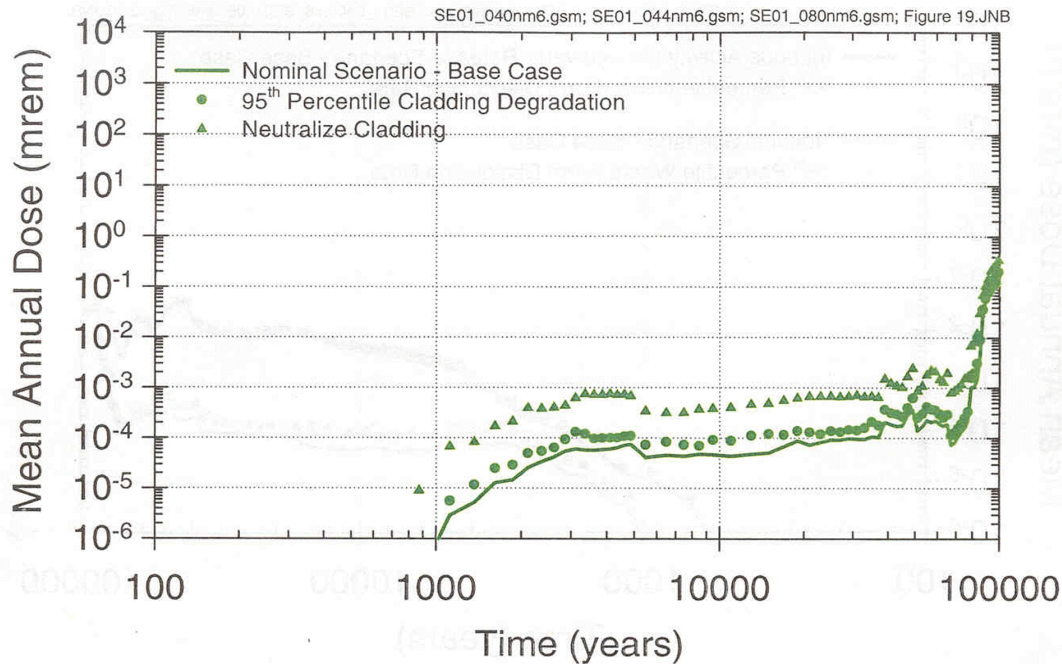
Figure 4-17. Sensitivity of Mean Annual Dose to the Commercial Spent Nuclear Fuel Cladding Degradation Rate (Bechtel SAIC Company, LLC, 2002. Note that Dose Estimates for Variations from the Basecase Do Not Represent Variations in Expected Risk because the Probability of the Variation Is Not Considered) (1.0 mrem = 0.01 mSv)

It is apparent in the TPA Version 4.1 Sensitivity Report (Figure 4-18) (Mohanty, et al., 2002) that the introduction of cladding protection decreases the dose at 10,000 years with respect to that for the basecase from 2×10^{-4} to 3×10^{-6} mSv/yr (2×10^{-2} to 3×10^{-4} mrem/yr). Release rates of highly soluble and mobile radionuclides like Tc-99 and I-129 account for most of the 10,000-year predicted dose, and are approximately proportional to the amount of spent nuclear fuel exposed. Other hazardous but less mobile radionuclides like plutonium and americium may not be as affected by the amount of spent nuclear fuel exposed, because their release is likely to be controlled by solubility limits.

Uncertainties

Cladding perforations may expose increasing areas of the spent nuclear fuel matrix to the in-package environment as a result of splitting (unzipping) caused by the oxidation of UO_2 pellets, either by air and water vapor or by liquid water. Perforations could exist in the cladding before waste emplacement and can develop by the corrosion and mechanical processes described above. Considerable uncertainties exist regarding the initial stage of cladding failure during which perforations occur, particularly in the case of corrosion processes. These uncertainties are mostly associated with uncertainties in the in-package environment and the initial condition of the spent nuclear fuel. A more significant range of uncertainties can be expected for high burnup fuel. Another concern is the potential for cladding failure under seismic conditions, rockfall, and drift collapse, with their associated uncertainties.
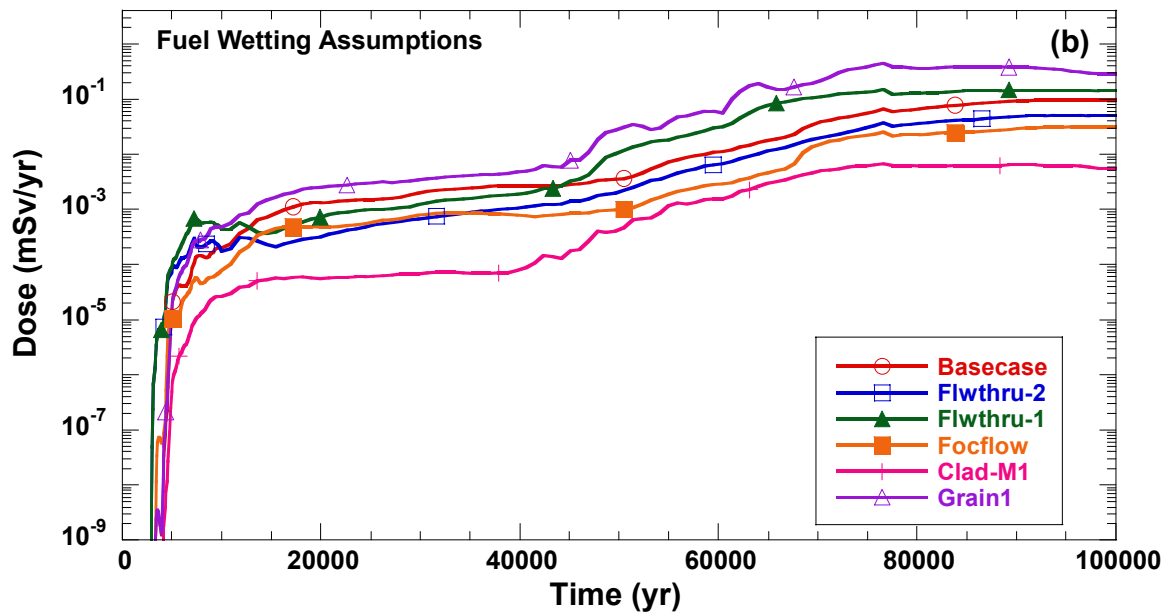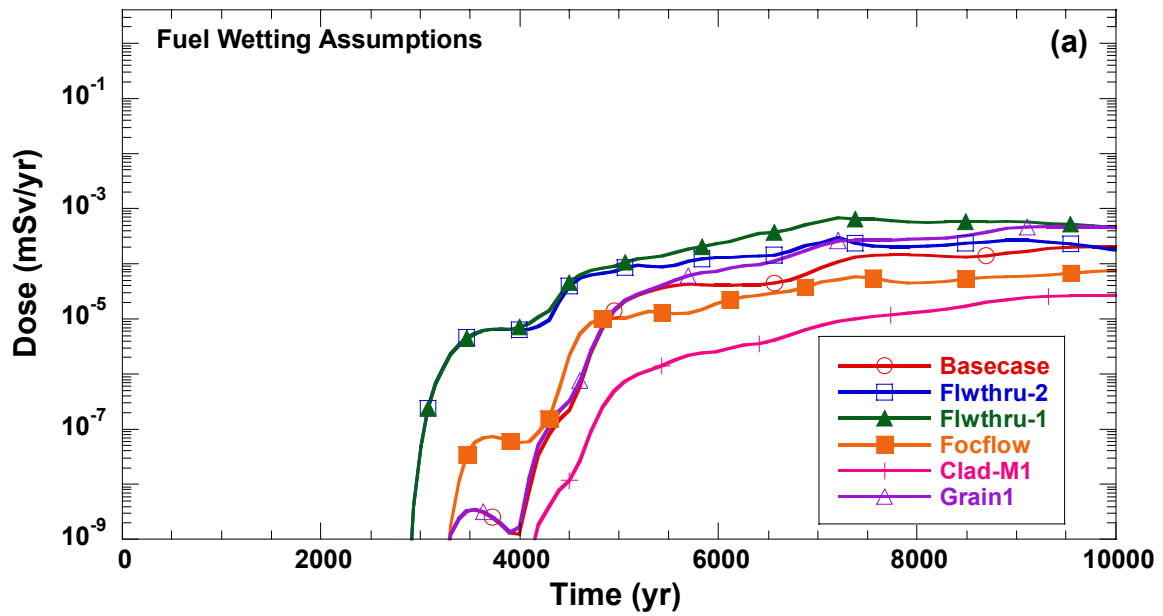
Figure 4-18. Effects of Cladding Credit: Ground Water Dose from the Basecase and the Fuel-Wetting Alternative Conceptual Models for (a) 10,000 Years and (b) 100,000 Years, Using the Mean Value Data Set (Mohanty, et al., 2002, Figure 3-37)

Solubility Limits: Medium Significance to Waste Isolation

Solubility limits effectively limit the release of many radionuclides.

Discussion

Solubility limits can be important factors in the release of radionuclides from the engineered barrier and ultimately to dose. Figure 4-19 shows the sensitivity of the DOE calculated doses to neptunium solubility. The degree to which radionuclide release to the environment is solubility-limited depends on the intricate interaction among the waste-form leaching rate, the degree of waste-form exposure to water, radionuclide half-life, radionuclide inventory, and the position of the radionuclide in the decay chain. Of the three radionuclides estimated to be major contributors to dose in a recent numerical study (i.e., Tc-99, I-129, and Np-237), only the release of Np-237 would be significantly decreased by its solubility limit (Mohanty, et al., 2003). However, 13 out of 20 radionuclides considered in this study exhibited solubility-limited behavior. Figure 4-20 plots the difference in cumulative release for each radionuclide from the waste packages between the basecase and a modified case in which the solubility limits for all radionuclides were set to an artificially high value. This figure shows that radionuclides such as americium and plutonium exhibit increased release when solubility limits are increased. Under the expected performance of other barriers, releases of radionuclides like plutonium and americium would not normally be seen because they are relatively immobile in the geosphere.

Uncertainties

The solubility value for particular radionuclides used in performance assessment calculations will depend on assumptions regarding the solubility controlling solid phase for the radionuclides of concern. The assumed solid phase may differ in its solubilities by several orders of magnitude. For some radionuclides, such as neptunium, incorporation into a low-solubility secondary phase is possible, but the evidence for this mechanism can be tenuous (Fortner, et al., 2003).

A necessary assumption in assigning solubility values pertains to the physical-chemical conditions of the system. The solubility of a radionuclide will depend on the composition of the aqueous phase as well as on its temperature and oxidation state. Inorganic and organic ligands that can form aqueous complexes with the radionuclide may be present. Complexation increases the amount of the radionuclide in the solution for elements such as uranium, neptunium, plutonium, and americium. Actinide solution chemistry in environmental waters would be dominated by hydroxide and carbonate complexation; thus, the solubility of actinide solids would be highly dependent on pH, aqueous carbonate concentration, and partial pressure of carbon dioxide gas. The solubility of some radionuclides depends strongly on their oxidation states.

In some cases, radionuclides released from the waste form may become sorbed to solid materials. This sorption could either hinder or enhance transport from the engineered barrier, depending on whether the substrates were immobile or mobile. In the former case, experiments with degrading spent nuclear fuel show a large fraction of plutonium and americium releases became attached to the container walls (Wilson, 1989). In the latter case, radionuclides that

(a) Igneous Activity Groundwater Release Scenario
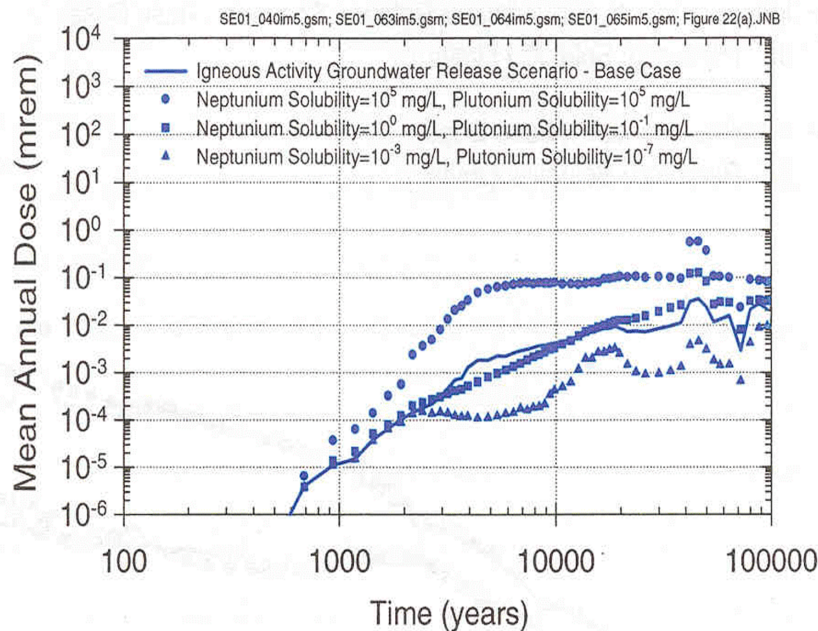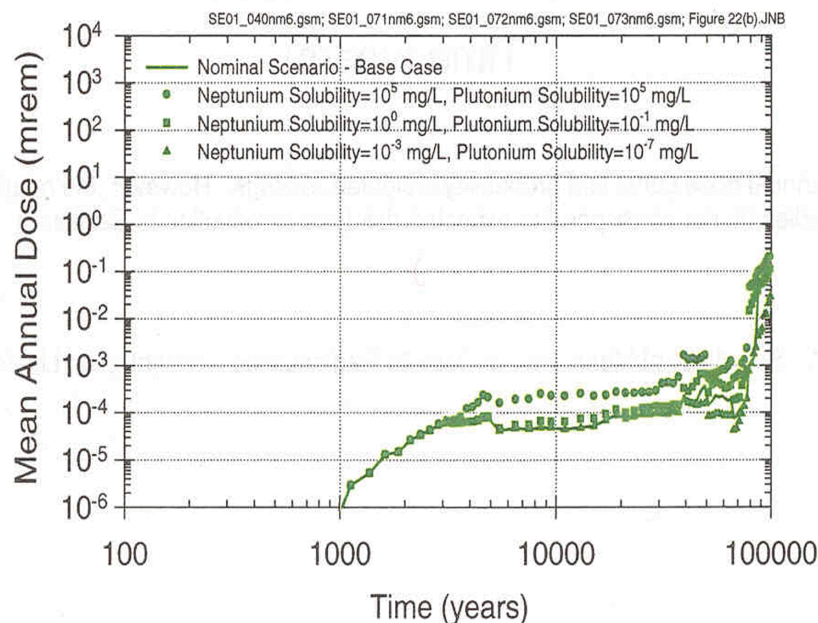
(b) Nominal Scenario

Figure 4-19.   Sensitivity of Mean Annual Dose to Concentration Limits for Neptunium and Plutonium (Bechtel SAIC Company, LLC, 2002. Note that Dose Estimates for Variations from the Basecase Do Not Represent Variations in Expected Risk Because the Probability of the Variation Is Not Considered) (1.0 mrem = 0.01 mSv)
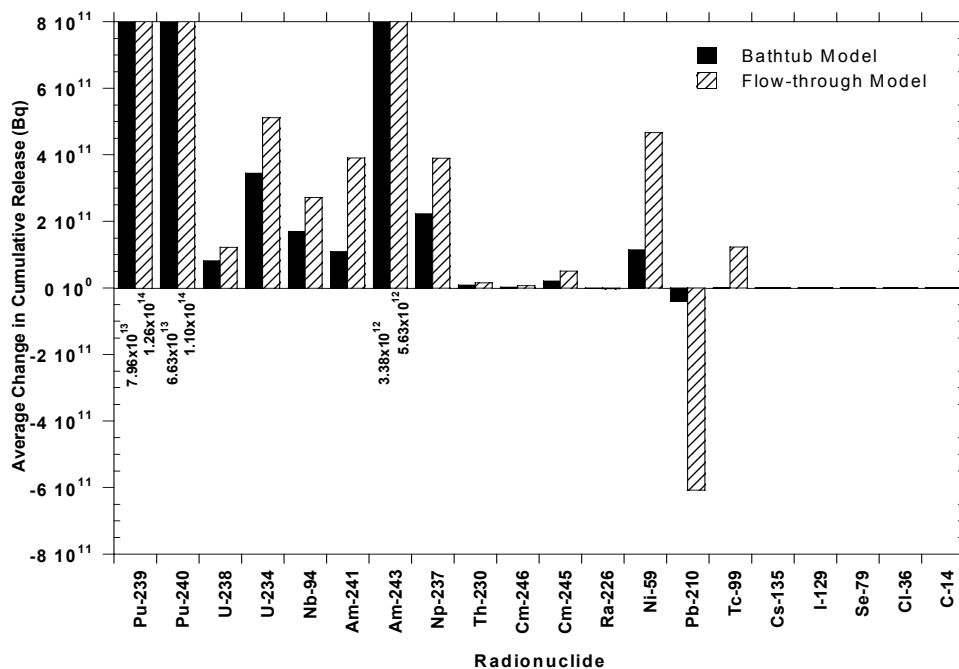
Figure 4-20. Average Change to the Nominal Scenario Cumulative Radionuclide Release in 10,000 Years When the Solubility Limit Is Specified as Ineffectivie in TPA Version 4.1 Code Simulations (i.e., Set to an Artificially High Value)[1]

become attached to colloids (e.g., hydrous ferric oxides generated by the corrosion of steel in the waste packages and ground support) might allow release of radionuclides in amounts greater than estimates based on solubility limits alone.

Mode of Release from Waste Package: Low Significance to Waste Isolation

The significance of diffusional release will depend on a number of assumptions such as water-film thicknesses, diffusion distances inside and outside the waste packages, unclogged openings, and a mechanism to sweep away contaminants, to keep the concentration gradients high. Advective releases rely on the quantity of water entering and leaving the waste package, and could be more significant than diffusion after degradation processes cause openings in the waste package that are sufficiently large to allow dripping water to come into direct contact with the waste. Processes leading to openings in the waste packages include localized corrosion, stress corrosion cracking coupled with mechanical loading events from dynamic and static rockfall loads, and intrusive igneous activity disruptive events.

Discussion

Scenarios capable of allowing diffusion in amounts that would be sufficient to cause significant releases are highly unlikely. There would have to be a continuous and substantial water pathway for diffusion and a mechanism to keep the concentration gradient high. Water films are likely to be thin or discontinuous. Under conditions where only water vapor is present in the drift, water films inside the waste package would be limited to layer thicknesses measured in tens of molecules or less. Mechanisms for flushing diffused radionuclides away from the waste

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[1]Mohanty, S., G. Adams, and R. Pabalan. "Effectiveness of Solubility Limit in Controlling Radionuclide Release from a High-Level Radioactive Waste Disposal Facility—A Numerical Study." *Nuclear Science and Engineering.* Submitted for publication. 2004.